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MEMORANDUM REPORT NO. 1438
NOVEMBER 1962

CALCULATION OF RADIATION DOSE LEVELS FOR THE ARMY PULSE EXPERIMENTAL REACTOR ASSEMBLY

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Terminal Ballistics Laboratory

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A B E R D E E N   P R O V I N G   G R O U N D,   M A R Y L A N D

B A L L I S T I C   R E S E A R C H   L A B O R A T O R I E S

MEMORANDUM REPORT NO. 1438

FHGregory/ic
Aberdeen Proving Ground, Md.
November 1961

CALCULATION OF RADIATION DOSE LEVELS FOR THE ARMY PULSE EXPERIMENTAL REACTOR ASSEMBLY

ABSTRACT

Calculations are made to estimate, as a function of distance, the radiation dose levels which would occur during full capacity operation of the Army Pulse Reactor Assembly (APRA) scheduled for construction at Aberdeen Proving Ground, Maryland. Calculations of the neutron dose are made by two methods. The first calculation (Method A) is made by utilizing an experimentally determined effective dose mean free path, which probably includes the effect of ground scattering. The second calculation (Method B) is made by utilizing the result of a Monte Carlo calculation of neutron transmission in air. The results of this calculation indicate considerably higher dose levels than indicated by Method A. Experimental data are available which substantiate the result of Method A. The gamma dose at distances of interest is found to be unimportant when considered in light of the uncertainty in the neutron dose.

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LIST OF FIGURES AND TABLES

I. INTRODUCTION

In order to determine the degree to which operation of the Army Pulse Reactor Assembly (APRA) at the proposed Aberdeen Proving Ground site will disrupt other nearby activities, it is necessary to calculate the radiation dose which will be delivered at various distances from the nuclear reactor.

In the case of an unshielded reactor such as APRA, the exclusion area will be determined by the dose produced by radiation emanating from the reactor. The proposed site is in the industrial area of the Proving Ground. Access to this area by the public and by all Aberdeen Proving Ground employees except those who have industrial area passes is restricted. The following proposals have been made to further restrict access to the reactor site and, at the same time, to provide adequate protection to insure the safety of persons not directly under the administrative control of the reactor staff.

a. There should be a barbed wire fence with appropriate signs attached at a distance such that the maximum weekly dose outside of this fence is 10 mrem.

b. There should be a security fence (6 or 8 foot fence with barbed wire barrier on top) at a distance such that the maximum daily dose outside of this fence is 10 rem.

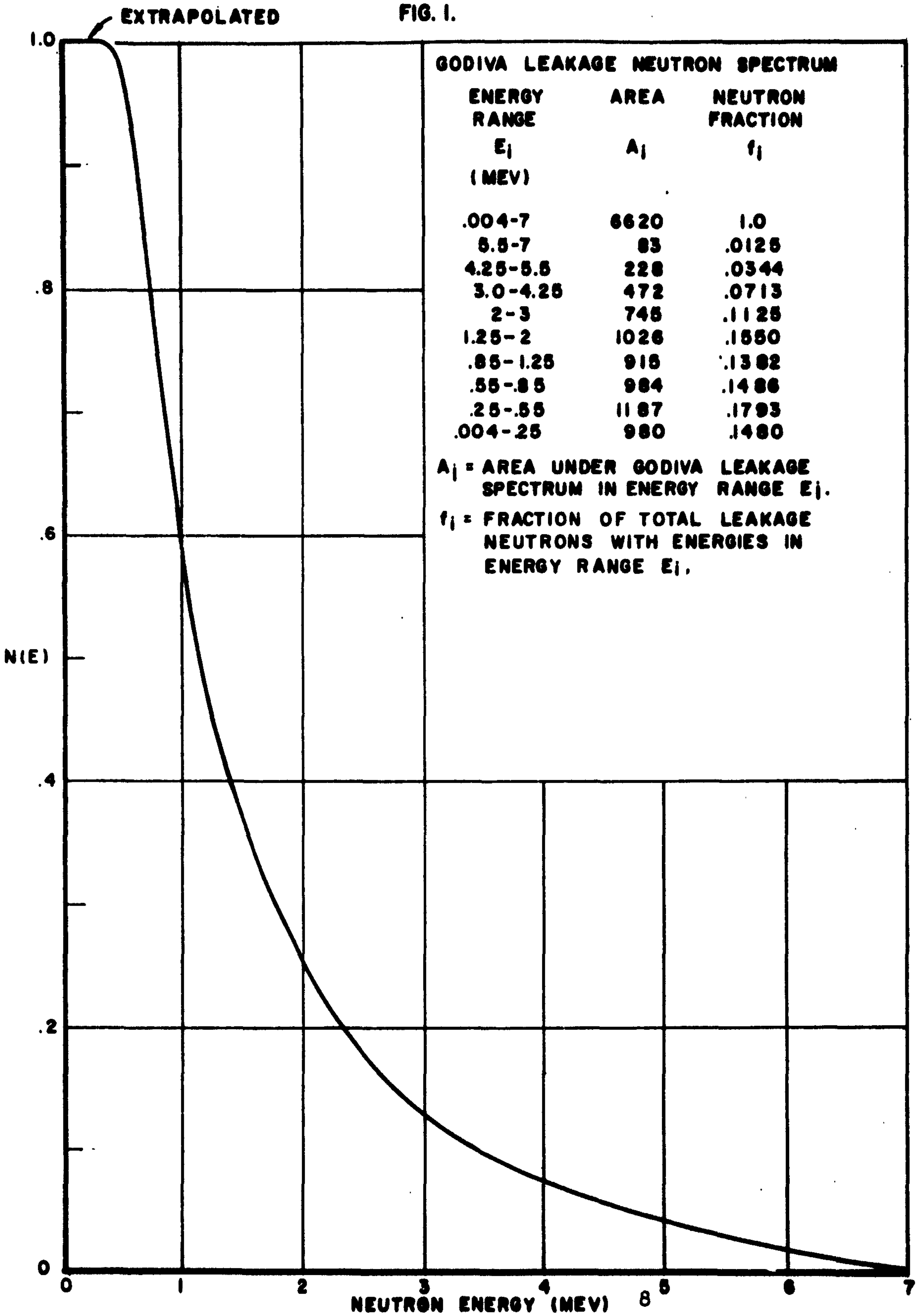
Members of the Hazards Evaluation Branch of the AEC have indicated that these restrictions would probably be considered adequate for exclusion of "non-occupational" personnel.

II. CALCULATION OF NEUTRON DOSE

The Godiva II neutron* leakage spectrum is used as a source model for the calculation by Method B. See Reference 1. This spectrum is included as Figure 1. The total source is assumed to be 80×10^{17} fissions, and the number of leakage neutrons per fission is assumed to be 1.4 (Reference 2, p. 9). This source is equivalent to 40 pulses per week at 2×10^{17} fissions per pulse. 2×10^{17} fissions is the probable maximum yield which the APRA will deliver in one pulse. The dose-distance calculation has been made by two different methods.

*The γ dose is estimated at about 2 per cent of the neutron dose at distances of interest. See Section III.

FIG. 1.



GODIVA LEAKAGE NEUTRON SPECTRUM

ENERGY RANGE E_i (MEV)	AREA A_i	NEUTRON FRACTION f_i
.004-7	6620	1.0
5.5-7	83	.0125
4.25-5.5	228	.0344
3.0-4.25	472	.0713
2-3	745	.1125
1.25-2	1026	.1550
.85-1.25	915	.1382
.55-.85	984	.1486
.25-.55	1187	.1793
.004-.25	980	.1480

A_i = AREA UNDER GODIVA LEAKAGE SPECTRUM IN ENERGY RANGE E_i.

f_i = FRACTION OF TOTAL LEAKAGE NEUTRONS WITH ENERGIES IN ENERGY RANGE E_i.

A. Method A

A simplified recipe for calculating the dose in air at several mean free paths from a neutron source has been proposed (Reference 3). The proposed method is as follows: (1) Use an infinite mean free path (no absorption) for the first 300 feet. (2) Use a mean free path of 212 yards from 300 feet on out. Changes in air density would cause, presumably, a change in mean free path.

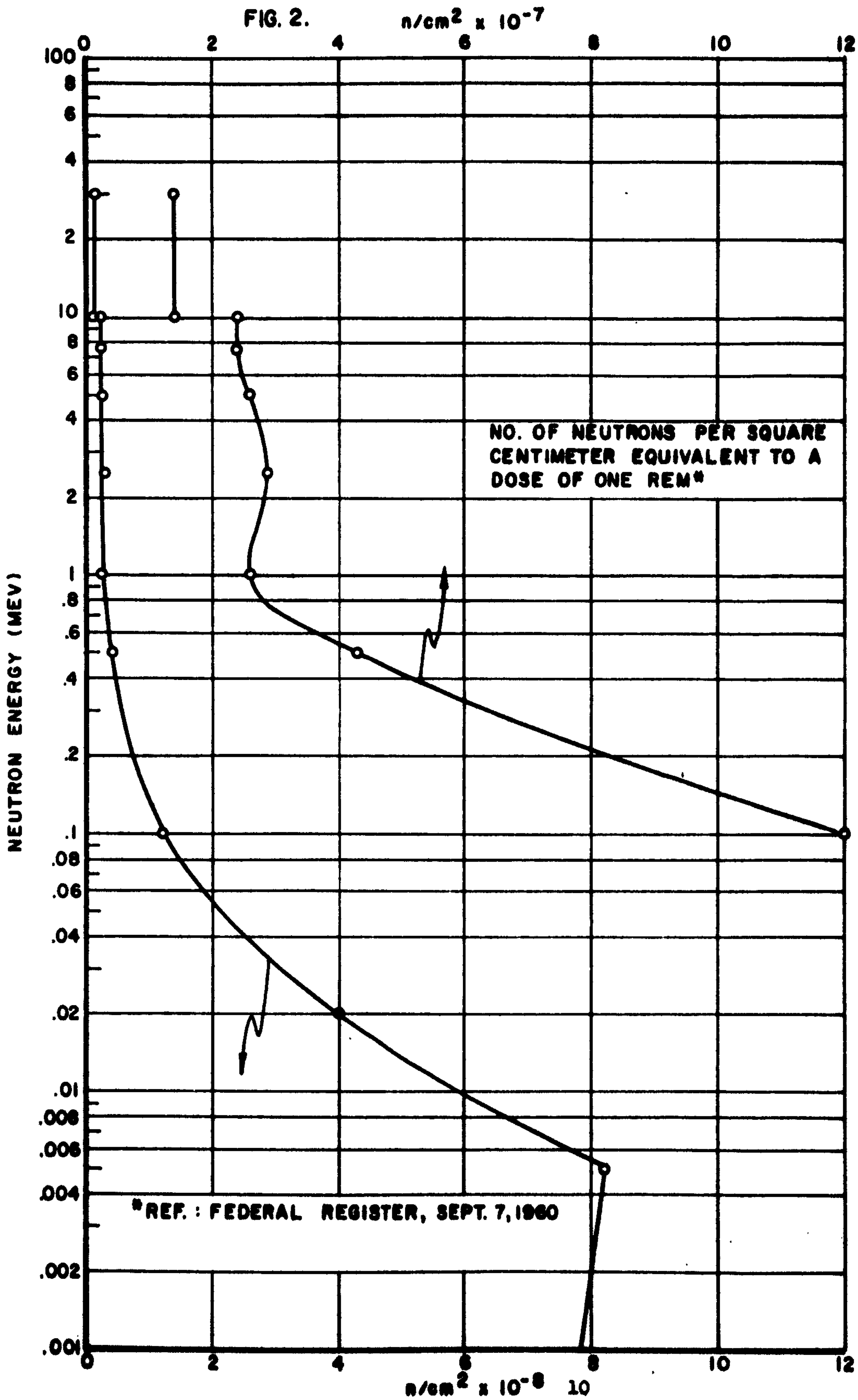
A calculation was made using the above method with an infinite mean free path for the first 100 meters. This method predicts 10 mrem/wk at 1110 meters (1215 yards).

B. Method B

A more complex calculation was made using the data presented by Mehl (Reference 4). Mehl gives dose-distance data in air for nine monoenergetic sources. The Godiva spectrum was approximated by building up a spectrum composed of the nine monoenergetic sources given by Mehl. An appropriate number of neutrons was assigned to each source according to the number of neutrons of the Godiva spectrum in the energy range corresponding to monoenergetic source. For each of the nine monoenergetic sources, Reference 4 gives the number of neutrons of a given energy at various distances from the source. For instance if one is interested in finding the dose received at 1000 meters from a source of 1 Mev neutrons, data are given to enable one to determine the number of neutrons in each of several energy ranges. The neutron dose at a distance R is determined by accumulating the number of neutrons arriving at R in each energy range from all source energies. The number of neutrons of a given energy required to make one rem is given in Reference 5. These data are presented in Figure 2. A sample calculation (with R = 1500 meters) is given on pages 6 and 7. The calculation is based on a total source of one neutron. Source neutron energies are distributed as indicated in Figure 1. Symbols used are defined as follows:

- i - subscript used to denote source neutron energy.
- ΔE_i - Energy range of the source neutrons
- F_i - Fraction of the source neutrons in energy range ΔE_i as determined from Figure 1.

FIG. 2.


n/cm² x 10⁻⁷
0
2
4
6
8
10
12
NEUTRON ENERGY (MEV)
100
8
6
4
2
10
8
6
4
2
1
.8
.6
.4
.2
.1
.08
.06
.04
.02
.01
.008
.006
.004
.002
.001
NO. OF NEUTRONS PER SQUARE CENTIMETER EQUIVALENT TO A DOSE OF ONE REM*
*REF. : FEDERAL REGISTER, SEPT. 7, 1960
n/cm² x 10⁻⁸ 10

R - Range (distance) from the source in cm.

j - Subscript used to denote neutron energy range at R.

ΔE_j - Energy range of the neutrons at R

N_{ij} - Number of neutrons per square cm at R in energy range ΔE_j which started with a source energy in ΔE_i (assuming total source of one neutron)

C_j - Conversion factor for determining the number of neutrons/cm^2 of energies ΔE_j necessary to make one rem.

D_j - Neutron dose at R due to neutrons with energies in energy range ΔE_j.

The air density used by Mehl is 1.07 grams/liter. The mean July air conditions for APG are as follows:

$$T_{avg} = 24^{\circ}\ C$$

$$B = 761.5 \text{ mm Hg}$$

$$\text{Rel. Hum} = 71 \text{ per cent}$$

These conditions indicate a mean July (when the monthly mean density is lowest) density of 1.181 grams/liter. A density correction is applied according to the following formulas:

$$\rho_1 R_1 = \rho_2 R_2$$

$$D_1(\rho_1 R_1) = (\rho_1/\rho_2)^2 D_2(\rho_2 R_2)$$

where ρ = air density

R = distance from the source

D = dose

In the calculation above, we have R_1 = 1500 meters, $D_1 = 2.42 \times 10^{-21}$ rem/source neutron, ρ_1 = 1.07 grams/liter, ρ_2 = 1.181 grams/liter. Using these values in the above equations, one gets: R_2 = 1359 meters, $D_2 = 2.95 \times 10^{-21}$ rem/source neutron (1.4 source neutrons/fission) = 4.13×10^{-21} rem/fission. Additional dose-distance data are shown in tabular form in Table II and in graphical form in Figure 3. The "inverse square loss only" curve is shown for comparison purposes only.

FIG 3. DOSE-DISTANCE CURVES FOR APRA

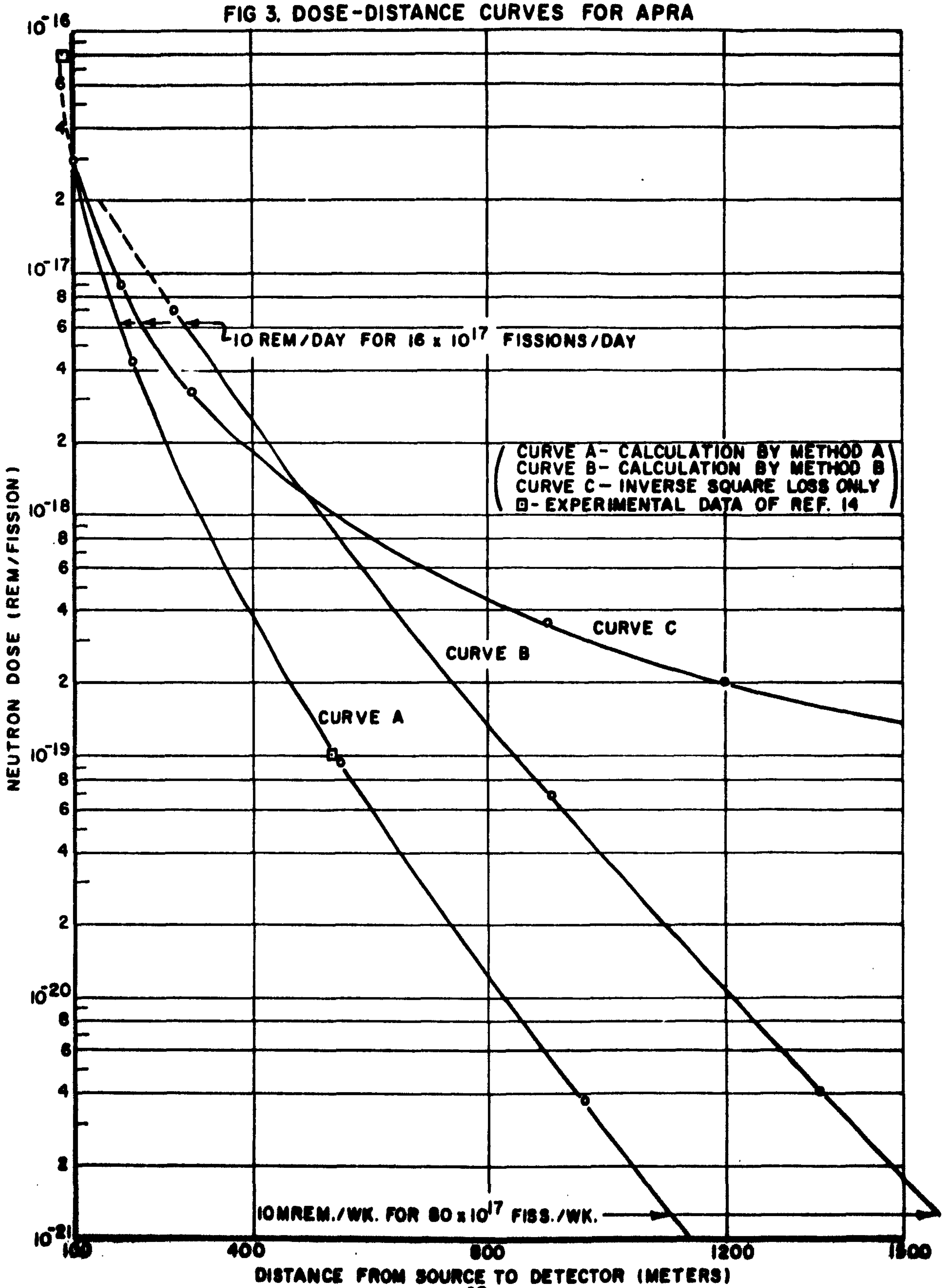


TABLE I

NEUTRON CALCULATION FOR 1500 METERS (METHOD B)

				$j \rightarrow$	1	2	3	4	5	6	7	8	9
				$\Delta E_j \rightarrow$	.0002 - .004	.004 - .1	.1 - .4	.4 - .7	.7 - 1.1	1.1 - 1.5	1.5 - 2.5	2.5-4.0	4.0
i	ΔE_i (Mev)	f_i	f_i/R^2	$N_{ij} \rightarrow$	N_{i1}	N_{i2}	N_{i3}	N_{i4}	N_{i5}	N_{i6}	N_{i7}	N_{i8}	N_{i9}
0	0 - 7.0	1.00	$\times 10^{12}$		$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$	$\times 10^{15}$
1	5.5 -7.0	.0125	.556		3.00	5.67	1.50	2.67	1.168	.723	1.407	.723	.261
2	4.25-5.5	.0344	1.529		5.81	11.93	3.06	5.66	2.98	1.835	2.75	.902	.245
3	3.0-4.25	.0713	3.169		10.77	19.33	3.64	8.30	3.20	1.838	3.169	.761	
4	2.0-3.0	.1125	5.00		13.50	27.75	6.50	10.25	4.75	2.250	3.50		
5	1.25-2.0	.1550	6.89		4.82	10.13	.59	1.72	.31	.052			
6	.85-1.25	1382	6.14		2.89	3.96	.50	.46	.04				
7	.55-.85	.1486	6.60		1.29	.94	.06	.05					
8	.25-.55	.1793	7.97		.08	.04	0						
9	.004-.25	.1480	6.58		.01	0							
				Totals	42.17	79.75	15.85	29.11	12.45	6.70	10.83	2.386	.506

TABLE I (Continued)

$\sum_i N_{ij}$ (n/cm^2)	C_j ($n/cm^2/rem$)	D_j (rem/n)
$x10^{15}$	$x10^{-7}$	$x10^{22}$
42.17	79.0	.534
79.75	33.0	2.417
15.85	7.8	2.032
29.11	4.0	7.278
12.45	2.74	4.544
6.70	2.64	2.538
10.83	2.84	3.813
2.386	2.78	.858
.506	2.6	.195

Total dose/n = 24.2×10^{-22} rem/n

C. Tabulation of results

TABLE II

Dose-Distance Data for Neutrons

Calc. Method	Range (Meters)	Dose (rem/fission)
A	960	3.75×10^{-21}
A	550	9.41×10^{-20}
A	200	4.31×10^{-18}
B	1359	4.13×10^{-21}
B	906	6.83×10^{-20}
B	272	7.00×10^{-18}
C*	1200	2.02×10^{-19}
C	900	3.58×10^{-19}
C	500	1.16×10^{-18}
C	300	3.23×10^{-18}
C	180	8.96×10^{-18}
C(and A)	100	2.90×10^{-17}

In addition to the above data there are some experimental data available from the Sandia Pulse Reactor. The following data are reported (Reference 14) for the reactor operated at the outdoor site.

Range (feet)	Dose (rem/fission)
250	8×10^{-17}
1750	5×10^{-20}

These data agree very closely with calculation A as shown in Figure 3.

*Calculated dose assuming no attenuation other than inverse square loss.

III. CALCULATION OF GAMMA DOSE

A. Fission Gammas

The total fission gamma source (delayed plus prompt) is assumed to be 13.1 Mev/fission. See Reference 6, p. 336. The distribution of gammas is taken as $n(E) = A \exp^{-cE}$ where $c = 1.16$ after Reference 7. By setting $A = c^2$, one normalizes the distribution to a total source of 1 Mev. Note that the average energy is

$$\bar{E} = \frac{\int_0^\infty E\, n(E)dE}{\int_0^\infty n(E)dE} = \frac{1}{c} \quad .$$

If $c = 1.16$, $\bar{E} = 0.86$ Mev. For prompt gammas, c is sometimes taken as $\simeq 1.4$ (e.g., Reference 7; also the estimated slope of the straight line drawn through data of Figure 2, Reference 6 is such that $c \simeq 1.4$). This gives $\bar{E}$ for prompt gammas of $\simeq .7$ Mev. However, Reference 6, p. 369 gives, for prompt gammas in the energy range .3 to 10 Mev, 7.2 ± 0.8 Mev/fission carried by 7.4 ± 0.8 photons/fission, or an $\bar{E}$ of $\simeq 1.0$. By manipulation of data presented in Reference 6, one finds that $\bar{E}_t \simeq 0.95$ Mev averaged over fission gammas emitted promptly, short delayed and long time delayed. If $\bar{E}_t \simeq 0.95$ Mev, the calculations given here indicate a gamma dose which is too small by 5 to 10 percent when one takes into account the fact that the nitrogen capture gamma dose is dominant at large distances. Since it will be shown that the gamma dose is small in comparison with the neutron dose for the situation under consideration, no further attempt will be made to resolve this apparent contradiction.

Using the distribution $n(E) = 13.1\, c^2 \exp.^{-cE}$, the energy range 0 to 8 Mev is broken down into energy groups of 1/2 Mev width as shown in Table III. The number of gammas in each group, n_{io}, is assigned the median energy of the energy range, E_i.

$$n_{io} = \int_{E_1}^{E_2} n(E)dE = 13.1\, c \left[\exp.\,(^{-cE_1}) - \exp\,(^{-cE_2})\right]$$

The number of photons in each energy group (n_{io}) is then attenuated according to the formula

$$n_i(\text{f.p.}) = n_{io} B_E(\mu_s Z, E) \exp(^{-\mu_s Z})$$

TABLE III

Calculation Of Photon Transmissions Due To Fission Products Decay And Neutron Capture By Nitrogen

i	Energy Range Mev	E_i Mev	μ_{mU} $cm^2/gm \times 10^2$	μ_{mMo} $cm^2/gm \times 10^2$	$\mu_s d$ cm^{-1}	$\mu_s Z$ cm^{-1}	B_E
1	0 - .25	-					
2	.25 - .75	.5	18.5	8.79	31.8	~ 4.75	1.49
3	.75 - 1.25	1.0	7.8	5.81	13.8	3.58	1.79
4	1.25 - 1.75	1.5	5.58	4.70	9.97	3.09	1.77
5	1.75 - 2.25	2.0	4.89	4.15	8.74	2.88	1.79
6	2.25 - 2.75	2.5	4.69	3.90	8.37	2.84	1.74
7	2.75 - 3.25	3.0	4.50	3.66	8.03	2.76	1.69
8	3.25 - 3.75	3.5	4.50	3.58	8.01	2.75	1.65
9	3.75 - 4.25	4.0	4.51	3.50	8.01	2.75	1.60
10	4.25 - 4.75	4.5	4.54	3.47	8.06	2.78	1.56
11	4.75 - 5.25	5.0	4.58	3.44	8.10	2.78	1.53
12	5.25 - 5.75	5.5	4.64	3.43	8.20	2.80	1.49
13	5.75 - 6.25	6.0	4.70	3.43	8.30	2.82	1.45
14	6.25 - 6.75	6.5	4.76	3.45	8.40	2.83	1.43
15	6.75 - 7.25	7.0	4.82	3.47	8.52	2.84	1.40
16	7.25 - 7.75	7.5	4.88	3.49	8.76	2.89	1.37
17	7.75 - 8.25	8.0	4.95	3.51	8.71	2.86	1.34
18	8.25 - 8.75	8.5					
19	8.75 - 9.25	9.0					
20	9.25 - 11	10.0					

TABLE III (Continued)

i	n_{i0}(f.p) No./fiss. $\times 10^2$	n_i(f.p.) No./fiss. $\times 10^3$	n_i(comp) No./fiss. $\times 10^2$	$r(1000,E_i)$ Mev/cm^3-Photon $\times 10^{16}$	$n_i r_i(1000)$ Mev/cm^3-fiss $\times 10^{18}$	$r(1500,E_i)$ Mev/cm^3-Photon $\times 10^{17}$	$n_i r_i(1500)$ Mev/cm^3-fiss $\times 10^{19}$	n_i(f.p.)$r(1000)$ Mev/cm^3-fiss $\times 10^{18}$
1	382.6							
2	500.5	64.6	6.46	.0082	.053	.0006	.004	.053
3	280.3	140	14.0	.047	.658	.0113	.158	.658
4	156.9	126.6	12.66	.121	1.532	.052	.658	1.532
5	88.0	88.5	8.85	.243	2.15	.140	1.239	2.150
6	49.6	50.0	5.00	.400	2.00	.270	1.350	2.000
7	27.5	29.4	2.49	.547	1.362	.474	1.180	1.362
8	15.5	16.4	1.64	.710	1.164	.700	1.148	1.164
9	8.63	8.84	.884	.896	.791	1.00	.883	.791
10	4.83	4.68	1.633	1.09	1.78	1.30	2.123	.510
11	2.71	2.57	.257	1.26	.324	1.61	.414	.324
12	1.52	1.38	5.671	1.45	8.223	1.95	11.06	.200
13	.85	.736	.074	1.65	.122	2.31	.171	.122
14	.49	.413	1.35	1.83	2.471	2.70	3.645	.076
15	.27	.221	.299	1.99	.595	3.06	.915	.044
16	.15	.114	.826	2.20	1.817	3.45	2.85	.025
17	.08	.061	.006	2.42	.015	3.90	.023	.015
18			.277	2.56	.709	4.40	1.219	
19			.131	2.77	.363	4.60	.603	
20			1.465	3.56	5.215	5.65	8.277	
				Totals	31.3×10^{-18}		37.9×10^{-19}	11.0×10^{-18}

where $B_E(\mu_s Z,E)$ is the energy buildup factor for a plane monodirectional gamma source in uranium obtained from Reference 8, p. 146. n_i(f.P.) is the number of photons due to fission products in energy group i which escape from the reactor. $\mu_s(E)Z$ is the "effective" number of relaxation lengths for the photon energy E. This factor takes into account the fact that many of the photons produced are absorbed in the reactor core itself. The relationship between the number of source radius relaxation lengths, $\mu_s(E)d$, and the "effective" number of relaxation lengths is given in Reference 9, p. 73.

The radius of the reactor core, d, is assumed to be 4.5 inches of which 7 percent (of the radius) is gaseous coolant channel which we will assume to be non-attenuating. These values were estimated from data given in Reference 10, p. 42. The reactor core is 90 percent U and 10 percent by weight Mo. The linear absorption coefficients are calculated by

$$\mu_s(E) = 0.93 \left[0.9\ \mu_{mU} + 0.1\ _{mMo} \right] \rho_{UMo}$$

where

μ_{mU} = the mass absorption coefficient for U,

μ_{mMo} = the mass absorption coefficient for Mo,

ρ_{UMo}, the density of the U-Mo, is taken to be 17.08 gm/cm^3 after Reference 11, p. 4. Values of the mass absorption coefficients $\mu_m(E)$ are taken from Reference 12. Thus the number of photons in each energy group escaping from the reactor is determined.

Nitrogen Capture Gammas

B. This source is directly dependent on the number of leakage neutrons per fission, η. The value of η is taken as 1.4 neutrons/fission (Reference 2, p. 9). The number of Mev of nitrogen capture (N.C.) gammas produced per fission, U_{NC}, is computed as follows:

$$U_{NC} = \eta \left.\frac{\sigma_N}{\sigma_{air}}\right)_{abs} \cdot \left.\frac{\sigma_\gamma}{\sigma_a}\right)_N \cdot \gamma_T \bar{E}_o$$

where $\left.\frac{N}{\sigma_{air}}\right)_{abs}$ = ratio of neutron absorption in nitrogen to that in air.

$\left.\frac{\sigma_\gamma}{\gamma_a}\right)_N$ = ratio of radiative cature (N.C.) to absorption in nitrogen.

γ_T = average number of photons/N.C.

$\bar{\bar{E}}_o$ = average energy of photons emitted in N.C.

The values used in this calculation are:

$$\left.\frac{\sigma_N}{\sigma_{air}}\right)_{abs} = 1.0$$

$$\left.\frac{\sigma_\gamma}{\sigma_a}\right)_N = \frac{.1}{1.84}$$

$$\gamma_T = \frac{7.53}{5.23} = 1.44$$

$$\bar{\bar{E}}_o = \frac{48.46}{7.53} = 6.44$$

$$U_{NC} = 0.71 \text{ Mev/fission}$$

The distribution of gammas emitted in neutron captures in nitrogen is given in Table IV (taken from Reference 7).

TABLE IV

Photon Emission Energies and Probabilities of Neutron Capture by Nitrogen

k	Energy E_k, Mev	Relative No. n_k	P_k
1	10.816	1.0	.01456
2	9.156	.09	.00131
3	8.278	.19	.00277
4	7.356	.56	.00815
5	7.164	.19	.00277
6	6.318	.90	.01310
7	5.554	1.5	.02184
8	5.287	2.3	.03349
9	4.485	.8	.01165
	Total	7.53	

$$\sum_{k=1}^{9} n_k E_k = 48.76 \text{ Mev}$$

$$P_k = \frac{n_k(.71)}{48.76} = \text{No. of photons of Energy } E_k \text{ per fission}$$

The number of photons, P_k, of energy E_k is then assigned to the median energy E_i of the particular energy group in which the energy E_k fails as indicated in Table III.

The nitrogen capture gammas are treated as a point source positioned at the reactor center. This approximation is justified on the basis that in this calculation we are only interested in the dose at 1000 yards or more and the great majority of all neutrons are absorbed well within this distance.

C. Composite Gamma Dose

The number of gammas in each of 19 energy ranges between 0-11 Mev, $n_i(\text{comp}) = n_i$, is determined by adding the contributions from fission gammas and N.C. gammas. The values of n_i are listed in Table III. Also listed in Table III are values of $r(x,E_i) = r_i$ which is the energy in Mev deposited per cm^3 in air at a distance x in yards from a source of one photon of energy E_i. Values of r_i are given in Reference 7 and are calculated by the formula

$$r_i = \frac{B_{ri}E_i\mu_i' \exp(-\mu_i x)}{4\pi x^2}$$

where B_{ri} is the dose buildup factor in air for photons of energy E_i, μ_i' is the linear absorption coefficient not including compton scattering component, μ_i is the total linear absorption coefficient including the compton scattering component, and x is the distance from the source in cm. The buildup factors are taken from Reference 7 as before. The air density for the gamma calculations is taken as 1.205 gm/l rather than 1.181 gms/l since data are already available at 1.205 gm/l. This small difference will result in very little inaccuracy of the final results.

From Table III, we see that

$$\sum_i n_i r(1000, E_i) = 3.13 \times 10^{-17} \text{Mev/cm}^3 \text{ - fission}$$

and

$$\sum_i n_i r\ (1500,E_i) = 3.79 \times 10^{-18} \text{ Mev/cm}^3\text{-fission}$$

Using the conversion factors,

$$\left(1.6 \times 10^{-6} \text{ ergs/Mev}\right) \left(\frac{\text{roentgen}}{87.7 \text{ ergs/gm}}\right) \left(\frac{1}{1.181 \times 10^{-3}} \frac{\text{cm}^3}{\text{gm}}\right) = 1.545 \times 10^{-3} \text{r/Mev/cm}^3,$$

one obtains the gamma dose per fission shown in Figure 4. For a 2×10^{17} fission pulse the gamma dose is $\sim$0.1 mr at 1000 yds and $\sim$ 0.01 mr at 1500 yards.

IV. ANALYSIS OF CALCULATION

A. Results

The gamma dose is less than 2 per cent of the neutron dose (as calculated by Method B) for distances out to about 1500 meters. Therefore, we may neglect the gamma dose.

The distance at which 10 mrem/wk will be accumulated from a weekly source of 80×10^{17} fissions is 1215 yards based on calculation Method A. The corresponding figure for Method B is 1700 yards.

The distance at which 10 rem/day will be accumulated is 195 yards by Method A and 315 yards by Method B.

B. Factors Bearing on the Results

It is believed that the results of Method A are more correct than the results of Method B. This contention is supported mainly by the experimental data presented. Factors having an effect on either or both of the two methods will be enumerated. The applicable calculational method is indicated by the letter (s) preceding.

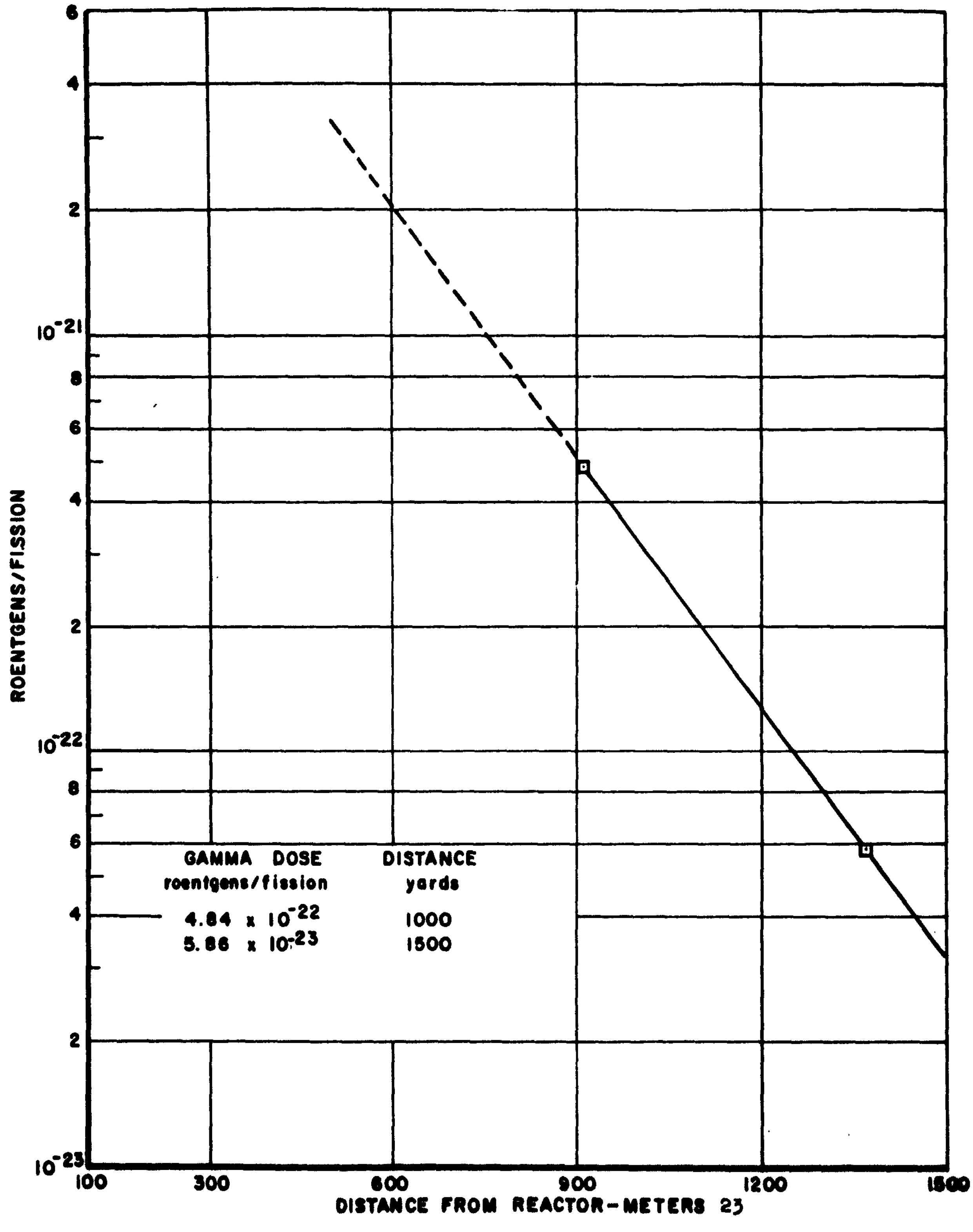
A,B 1. The radiation leakage from the APRA may be more (or less) per fission than that for Godiva (which was used in these calculations).

B 2. Thermal and resonance neutrons arriving at the detector below 200 ev energy have not been accounted for in this calculation.

A,B 3. The APRA neutron leakage spectrum may not be the same as the Godiva spectrum (which was used in these calculations).

B 4. This calculation is made for an infinite medium of air with a point source of neutrons (and gammas). The ground-air interface may decrease the dose at large distances from the reactor particularly when the reactor is operated close to the ground. Data are available for gammas

FIG 4. DOSE-DISTANCE CURVE FOR GAMMA RADIATION FROM APRA



(Reference 13, p. 89); however, no data has been found for neutron transmission. The non-nitrogen capture gamma dose will probably be reduced by about a factor of 2 or 3 at a distance of 1500 meters. It would not necessarily follow that this same figure can be used for neutrons, however.

A,B 5. It may not be necessary to pulse the reactor once per hour. Nevertheless, it would be desirable to be able to do so.

A,B 6. The APRA may not be capable of producing a pulse which is 13 times as large as the nominal Godiva II pulse of 1.5×10^{16} fissions/pulse.

B 7. Excessive moisture in the air will reduce the dose level below that calculated here. Mehl's calculations assume a constant air density of 1.07×10^{-3} gm/cm^3 with relative atom fractions of nitrogen, oxygen, and hydrogen of 0.7778, 0.2135 and 0.0085 respectively. APG air contains more than twice the amount of moisture as that used in Mehl's calculations (assuming mean July conditions for APG, the atom fraction for hydrogen is .0209).

B 8. The data used to calculate the dose by Method B are estimated to be accurate to within $\pm$ 50 per cent according to the author of Reference 4 (assuming an infinite medium of air).

A,B 9. Any attenuation of neutrons in the walls of the reactor building or experimental equipment positioned around the reactor will reduce the dose at distances of interest.

Of the above items, perhaps only the last one may be expected to result in a significant change in the predicted dose.

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BRL Memorandum Report No. 1438 November 1962

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